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AN EVALUATION PROGRAM
FOR A
RADIOACTIVE LIQUID WASTE
TREATMENT FACILITY

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U. S. ARMY CHEMICAL CORPS RESEARCH & DEVELOPMENT COMMAND
Nuclear Defense Laboratory
Army Chemical Center, Maryland

EVALUATION PROGRAM FOR A RADIOACTIVE LIQUID WASTE TREATMENT FACILITY

I. INTRODUCTION.

A facility for the concentration of low level radioactive liquid waste has been designed and fabricated for this Laboratory. Prior to releasing this facility for routine operation, an evaluation program will be conducted. The objectives of this evaluation are to:

1. Determine the optimum operating conditions of the radioactive liquid waste treatment facility.
2. Determine the capabilities and limitations of this facility.
3. Determine the effect of the operating variables on the decontamination factor (DF) and concentration factor (CF).

This work was authorized under USA CmlC R&D Project No. 4X12-01-001.

II. BACKGROUND.

A. Historical.

The Department of the Army has assigned responsibility for radioactive waste disposal to the USA Chemical Corps. One aspect of this responsibility is the disposal of large volumes of low level liquid waste. The Nuclear Defense Laboratory (NDL) has been requested to study and recommend disposal methods for these wastes.

A study of the problem by NDL indicated that a great variety of low level liquid wastes are generated at widely scattered locations. These include primary coolant water and decontaminating wastes generated at all U. S. Army reactor sites, and laboratory and laundry wastes generated by research laboratories using radioactive isotopes. These wastes may contain almost any isotope, along with corrosion products, acids, detergents, etc. Consequently a flexible, medium capacity, semi-fixed facility is needed.

B. Equipment Requirement.

The following general requirements were established for a unit to accomplish the task:

1. A capacity of 150-300 gallons per 8 hour work shift, including startup, shutdown, and sludge removal.

2. A minimum decontamination factor (DF) of 10^7 , feed to effluent.

3. A compact, flexible, semi-fixed design.

A contract was let to the Pfaudler Permutit Co. for the Unit. Under terms of this contract, the waste treatment facility will be delivered and installed at the Army Chemical Center during October 1961. Initial operation will be under the direction of NDL and will include a detailed evaluation of the unit's performance. The evaluation program is the subject of this memorandum.

C. Equipment Description.

The facility is an assembly of equipment mounted on a permanent steel frame to allow it to be moved readily to any site. It is complete in itself, requiring only process and utility connections for operation as a treatment system.

The facility includes a feed pump for effecting the transfer of stored wastes from existing storage tanks to a feed tank in which recycled residue or concentrate is mixed with dilute fresh feed. The mixture flows by gravity from the feed tank to a Pfaudler Wiped Film Evaporator, where condensing steam vaporizes part of the feed. Unvaporized residue drains into a residue tank, from which it is pumped back to the feed tank or to a sludge receiving drum. The vapors pass through an entrainment separator in the evaporator and an external separator to a condenser. Condensed vapors or distillate drain into either of two receivers. From this receiver the distillate is pumped through either or both of two mixed bed demineralizers to either of two final monitoring tanks. After this final check point the distilled and/or demineralized water is discharged from the facility.

The facility is designed to be vapor tight, but is expected to be operated under a slight negative pressure to prevent any out-leakage. The negative pressure will be produced by a fan or blower to which the vent line is connected.

Sample or drain points are provided so that activity checks may be made of the solution at any point in the facility and so that it can be completely drained.

Provision is also made to recycle solution through any component, even back to the main storage tanks, but no provision has been made for by-passing the evaporator.

D. Operational Description.

Figure 1 is a flow diagram of the Waste Treatment Facility. The waste water from storage tanks external to the unit is drawn through

a 3/4-in. feed line and line 3/4 A-1 to the self-priming feed pump, P-1. This pump then elevates it, through 1/2 A-2 to the feed tank, T-1. Rate of flow to T-1, as indicated by the rotameter, FI-1, is held to a maximum of about 1 gpm by a manually positioned valve. The flow is further reduced by control valve, LCV-1, which opens on low level in the concentrate receiver, T-2, and closes on high level. This control prevents evaporation to dryness or flooding of the system by adjusting the fresh feed rate to maintain a constant level of solution in T-2.

Concentrated solution is pumped from T-2 to T-1 at a nearly constant rate by P-2. It overflows from a 2-in. pipe inside of T-1 to mix with fresh dilute feed. The mixed solution flows by gravity from T-1 to E-1 through a venturi flow indicator, FI-2, and around a pH electrode. The pH of the solution is recorded on the control panel. Should this value be too low, the operator will add caustic solution from a bottle feeder, T-5, to T-1.

Solution flows down the inside of the steam heated walls of the evaporator, E-1. Residue drains from E-1 to the Concentrate Receiver, T-2, and water vapor passes through the internal separator of E-1 to the external separator, S-1.

Specific gravity of the concentrate is measured by bubbling air slowly into the bottom of the 2-in. concentrate overflow pipe of T-1. The air pressure required to obtain bubbling is equal to the hydrostatic head which, because of the fixed liquid leg, is a direct measure of specific gravity. Specific gravity measured in this way is recorded by DRA-1.

When the specific gravity of the concentrate reaches a pre-determined value, the operator will take a sample. If on cooling to 100° F the sample solidifies, then the flow of concentrate from T-2 to T-1 is diverted manually, in whole or in part, to a 55 gallon drum.

After removing sufficient concentrate to lower its specific gravity sufficiently as shown by DRA-1, the concentrate drain valve to the drum is closed.

This removal of concentrate or sludge is done without interruption of the evaporating process. The record shown by DRA-1 will be one of slow increase in specific gravity during concentration with an abrupt decrease when sludge is removed.

Water vapor produced in the evaporator passes through two internal separators and exits to the external separator, or demister, S-1. Any liquid which collects in S-1 drains back into T-2. From S-1 the vapor goes to the condenser C-1.

Water condensed in C-1 drains into either of the two distilled water receiver tanks, T-3 or T-3A. A 3-way air operated valve, LSV-1, directs the flow to either T-3 or T-3A. A probe type level switch in T-3 and in T-3A is the primary sensing device which controls the air supply in LSV-1. These will operate to change the direction of flow from the tank being filled to the other on high level. An alarm will sound when the switchover takes place. A manual push button will be required to turn off the alarm.

After a receiver tank, T-3 or T-3A, has filled and LSV-1 has redirected the flow of distilled water to the other tank, the operator will start the distilled water pump, P-3, to pump the water through the demineralizers.

Valving will provide complete flexibility in the use of the two demineralizers, IX-1 and IX-2. It will be possible to use either unit alone or in either of the two possible series connections.

Demineralized water is collected in either of the two 300 gallon Monitoring tanks T-4 and T-4A. As in the case of the distilled water, a 3-way air-operated valve will divert the demineralized water to the other tank and sound an alarm each time a tank fills to the level of the sensing probe. After the activity of the demineralized water collected in a monitoring tank has been checked, the contents of the tank are pumped by P-4 either to waste or back to T-3 or T-3A.

III. PROPOSED PROGRAM.

A. Discussion.

Prior to release of the facility to the U. S. Army Chemical Corps Materiel Command for routine operation, the Nuclear Defense Laboratory will evaluate the unit to fulfill the objectives outlined in section I and to train the future operating personnel. Since this will be the first time a wiped film evaporator has been used for radioactive waste treatment, data obtained during this evaluation program will permit comparison of efficiency and capability with standard evaporators.

In addition, experience gained during the program will assist in the completion of an operational manual and will provide a basis for future recommendations in radioactive liquid waste disposal.

During the evaluation program, the unit will be operated for one shift only. The unit will be started up in the morning, operated for 5 to 6 hours, and shut down. One day's operation will constitute a run. Data and samples will be taken after the unit has reached steady state (estimated at less than 1 hour). A sample data sheet for one run is included as Appendix A. It is estimated that the test program will be completed in six months.

The accumulated wastes at the Army Chemical Center will be treated during the test program. Preliminary analyses of these wastes indicate that the activity level ranges from 10^{-2} to 10^{-5} $\mu\text{c/ml}$, the pH is between 4 and 7 and the total solids content is 0.05% to 0.5%. It is expected that similar additional waste will be received during the test period. Since these wastes come from decontamination operations, laboratories, and laundries, almost any nuclide may be present.

Federal Register Part 20, "Standards for Protection Against Radiation", permits 1×10^{-8} $\mu\text{c/ml}$ of radiation of unknown radiochemical composition to be discharged into water in an unrestricted area. If specific nuclides are known to be absent, higher activity levels may be discharged. In particular, absence of Ra^{228} and Ra^{226} will allow discharge of 1×10^{-7} $\mu\text{c/ml}$ activity. In addition, if Pb^{210} and Sr^{90} are absent, 6×10^{-7} $\mu\text{c/ml}$ may be discharged, and so on. If, during the operation of the waste treatment facility it is found that the effluent from the evaporator has an activity level of $1-6 \times 10^{-7}$ $\mu\text{c/ml}$, a determination of Pb^{210} , Ra^{226} , Ra^{228} , and Sr^{90} might eliminate further processing. The availability of relatively quick and simple analytical means for determining these nuclides would conserve ion-exchange resin and reduce operational cost. Appendix B discusses available analytical techniques for determining the concentration of the hazardous nuclides.

During this evaluation the decontamination factors for the evaporator and ion-exchange units will be investigated to assure proper performance under a variety of solution compositions. The decontamination factors for available waste solutions can be determined by alpha and beta counting of an evaporated sample of effluent. In order to perform a more significant evaluation, it would be worthwhile to spike the waste solution with those nuclides having the most stringent decontamination requirements, (i.e., Sr^{90} and Ra^{226}). It will not be necessary to evaluate Pb^{210} since it is most improbable that this isotope will be found in Army use. Pb^{210} is found in the mining industry, whereas the Army works with fission products and induced activities. Also, it is desirable to spike the waste with nuclides that will most probably be present during routine treatment. For example, Co^{60} is prominent in the samples of waste from the SM-1 reactor at Fort Belvoir, Virginia. To fully evaluate the performance of the equipment with these materials, it would be desirable to analyze the effluents from each stage for the residual carry-through of these nuclides. For this type of evaluation, analytical procedures that have been laboratory tested are necessary. A complete discussion of the available procedures and recommendations is included in Appendix B.

B. Operating Variables.

1. Feed Rate and Recycle Ratio to Evaporator.

Initial runs will determine the effects on the DF and CF of varying the feed rate and recycle ratio. The feed rate is determined

by the amount of liquid distilled in the evaporator, and actually can be varied only by varying the steam input to the evaporator. Runs at reduced capacity will be used to determine the DF and CF as a function of capacity.

The recycle ratio will be controlled by a throttle valve in the recycle line. Special efforts will be expended to evaluate the effect of the recycle ratio, since it can be controlled easier than the feed rate. It is possible that the effects of the feed rate and recycle ratio on the DF and CF will be small.

2. Density of sludge solution.

When the hydrostatic head in the central overflow pipe of the feed tank reaches a predetermined value, the concentrated residue is removed. At this predetermined hydrostatic head, which is directly proportional to the density of the sludge solution, a valve is opened by the operator and the sludge flows into the sludge drum. The higher the sludge density, the greater is the solids content of the sludge and the higher the CF. The sludge density can probably be raised until the solids content becomes so high that the evaporator and recycle pump do not function properly. The DF may also be adversely affected by a high solids content in the sludge. The sludge density will be an important variable and will be evaluated thoroughly.

3. Feed Composition to the Evaporator.

Since the evaporator is the primary component of the facility, the composition of the feed to the evaporator is a major variable. Therefore, the capabilities and limitations of the unit will be measured with a wide variety of feeds. The ability of the unit to handle solutions that tend to foam will be determined. The DF of chosen nuclides, such as Ra^{226} , Sr^{90} , and Co^{60} will be determined by "spiking" feed solutions with these isotopes. Primary coolant water will be stimulated and processed. Other properties, such as activity level, pH, solids content, and temperature will be included in this phase of work.

4. Feed Composition to the Ion Exchanger.

The evaluation of the effect of pH, temperature, etc. should be rather simple and can be evaluated simultaneously with other variables. Much of the information may be available from the manufacturer of the ion exchange resins.

5. Type of Ion Exchange Resin.

The Waste Treatment Facility will contain two ion exchange units. Initially, one of these units will contain standard mixed-bed resins and the second one will contain a new resin which is used specif-

ically for Co⁶⁰. Both units will contain resins manufactured by the Permutit Division of Pfaudler Permutit, Inc. The use of these units alone and in series will also be evaluated.

6. Steam Temperature to Evaporator.

The upper limit of the steam temperature in the evaporator is determined by the pressure limitation on the steam jacket (i.e., Pressure = 75 psig, Temperature = 320° F). The lower limit is determined by the boiling point of the feed solution (probably around 250° F). Consequently, the steam temperature can be varied only within narrow limits. However, The Pfaudler Company states that the steam temperature affects the DF in the wiped film evaporator. This may not be an important variable, and should be readily evaluated.

7. Time and Use.

The effects of time and use will be difficult to evaluate during a six-month test period. Periodic calculation of the overall heat transfer coefficients in the evaporator and condenser will give an indication of scale buildup in the equipment. Solids and activity accumulation in any of the equipment will indicate weak points of the system. Repeating some of the initial runs near the end of the six-month test period will determine if the concentrator is retaining its efficiency. The evaporator and any questionable equipment will be disassembled at the end of the test period for inspection.

8. Additional Comments.

The amount of time devoted to the evaluation of each variable will depend on the importance of that variable as determined by the first few runs. The more important ones will receive the most attention. As stated previously, it is estimated that a major portion of the program will be varying the feed solution.

The efficiency of the unit at each set of operating variables will be evaluated by calculating the DF's of the evaporator, demister, and ion exchange units, and by calculating the concentration factor. The DF of the demister can be estimated by measuring the activity and volume of the liquid in the drain line from the demister. The heat transfer coefficients in the evaporator and condenser, and activity buildup in each piece of equipment, will also be measured to determine the efficiency of operation of the unit. All of these values can be calculated from the data taken during a day's run. A sample data sheet for a day's run is included as Appendix A.

An anticipated difficulty in the test program is the gathering of representative samples from various tanks and lines. Most of the calculations depend on the sample analyses. During the first few weeks of

operation, four or more samples will be taken during the day. If these samples differ appreciably in their analyses, the tanks may have to be agitated during the test program to insure adequate mixing.

Another difficulty that may be encountered is the control and separation of the variables. However, as experience is gained in operating the equipment, this difficulty should be eliminated.

It is anticipated that the evaluation program will require six months to complete.

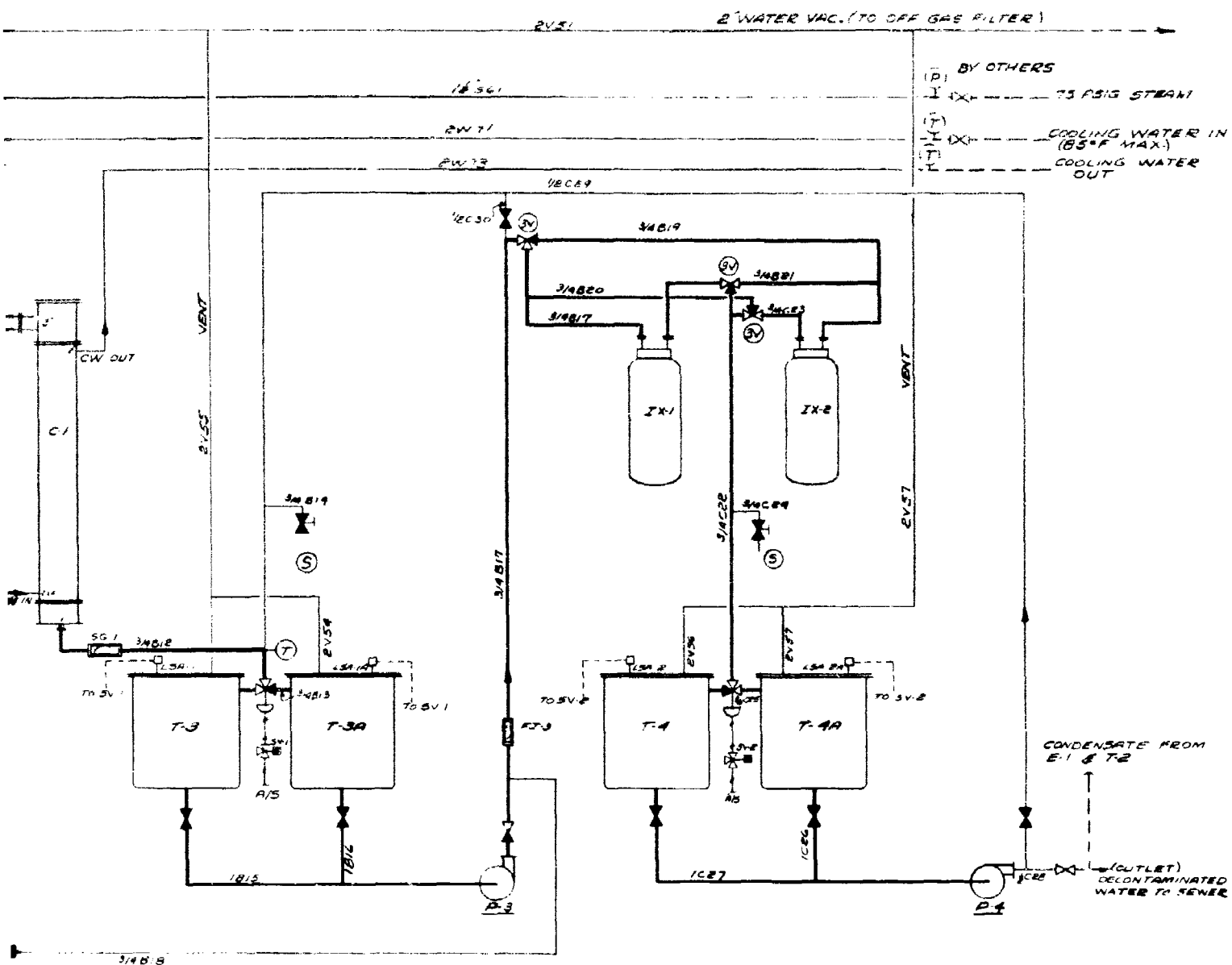


FIGURE 1

FLOW DIAGRAM
PACKAGED RADIOACTIVE WASTE CONCENTRATOR

REV.	DATE	BY	CHKD.
1	10/1/70	J. B. G.	J. B. G.
2	10/1/70	J. B. G.	J. B. G.
3	10/1/70	J. B. G.	J. B. G.
4	10/1/70	J. B. G.	J. B. G.
5	10/1/70	J. B. G.	J. B. G.

REFERENCE	USED ON
G-761/1200	



THE PFAUDLER CO.
A DIVISION OF PFAUDLER PERMUTIT INC.
ROCHESTER 3, NEW YORK
MADE IN CLEVELAND, OHIO ROCHESTER, N.Y.

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DATA SHEET

I. Startup.

C. Volume of Liquid in Tanks:

T-3A T-4 T-4A

E. Weight of Sludge Drum _____

[illegible]

III. Steady State Operation.

Date _____

A. Estimated time steady state reached _____.									
Time:									Comments
B. Steam									
1. Temp									
2. Press									
C. Steam Condensate									
1. Temp.									
2. Flow Rate									
D. Cooling Water									
1. Temp. In									
2. Temp. out									
3. Flow Rate									
E. Feed and Recycle Rate									
1. Feed Rate (FI-1)									
2. Feed and Recycle (FI-2)									
3. Temp. (TI-2)									
4. Recycle Rate									
5. Recycle Ratio									
F. Residue									
1. Temp. (TI-1)									
2. Wt. Discharged									

Appendix A

IV. Samples.

Date _____

No.	Location	Station	Temp	Time		Comment
				Start	Finish	
	Storage Tank					
	Feed Tank	T-1				
	Recycle Tank	T-2				
	Demister Drain	S-1				
	Tank					
	Tank					
	Vent					

V. Ion Exchange Operation.

From Tank	To Tank	Via	Time		Flow Rate (gpm)	Comment
			Start	Finish		

VI. Condensate Release.

A. Volume _____ B. From Tank _____ C. Time _____
 D. Sample # _____ E. Act. level _____ F. Tot. Act. _____
 G. Isotopes present _____

VII. Shutdown.

A. Time _____ B. Entrainment Volume _____
 C. Volume of liquid in tanks:
 T-1 _____ T-2 _____ T-3 _____
 T-3A _____ T-4 _____ T-4A _____
 D. Electric Meter Reading _____
 E. Weight of Sludge Drum _____

VIII. Radiation Dose Rate at Surface of:

1. Feed Tank (T-1) _____	8. T-3A _____
2. Recycle Tank (T-2) _____	9. T-4 _____
3. Evaporator (E-1) _____	10. T-4A _____
4. Sludge Drum _____	11. IX-1 _____
5. Demister (S-1) _____	12. IX-2 _____
6. Condenser (C-1) _____	13. Vent Line Filter _____
7. T-3 _____	

APPENDIX B

PROPOSED RADIOCHEMICAL ANALYSES

I. ANALYTICAL METHODS.

A search of available literature was undertaken to locate analytical procedures for Ra^{228} , Ra^{226} , Co^{60} , and Sr^{90} . The various methods in the literature were reviewed and if more than one was available, the method which seemed shorter, most applicable, and requiring a minimum of elaborate equipment was selected.

A. Radium-226.

Radium-226 is usually determined by counting its 3.8 day radon daughter by the emanation technique. This technique is the most sensitive, but requires elaborate equipment and considerable time. A number of methods have been published for the determination of total radium (Ref. 1,2,3). The method chosen was selected on the basis of simplicity, time required, and its designed application. Baratta and Harrington (Ref. 4) report an analytical procedure for total Ra applicable to United States Federal Register, Title 10, Part 20 levels or below, although it does not have the inherent accuracy of the emanation technique for Ra^{226} . The method was devised for application to uranium mill effluents but should be applicable to liquid waste. Basically the method determines total radium by coprecipitating radium with barium sulfate in the presence of EDTA. The precipitate is metathesized with sodium carbonate. The barium-radium carbonate is dissolved in nitric acid, and the resulting solution buffered to pH 4 with sodium acetate-acetic acid. The solution is then extracted with 2-thenoyltrifluoroacetone (TTA) in benzene to remove last traces of contaminants. The aqueous layer is treated with sulfuric acid to precipitate barium-radium sulfate. The precipitate is placed on a planchet, ignited, weighed, and counted for alpha and beta to give total radium activity and chemical yield.

B. Radium-228.

Radium-228 is a beta emitter and it is more difficult than Radium-226 to establish its presence. The radium method outlined will carry to the final precipitation 44% of any actinium present in the original solution. Actinium is also a beta emitter. Therefore, beta counting the final precipitate will not definitely establish the Ra^{228} content. It could be assumed that any beta activity is from Ra^{228} and proceed with a decision to pass the evaporator effluent through the ion-exchange on this assumption. An estimate of the time required to run four samples for total radium is about 4 hrs.

A method for determining Ra^{228} has been reported by the Winchester Laboratory (Ref. 5). This procedure is dependent upon the determination of Ra^{228} daughter actinium. Purified barium-radium sulfate is stored for approximately 48 hours. The resulting Ac^{228} from decay of Ra^{228} is reported by means of an actinium procedure. Actinium is carried on lanthanum fluoride, purified by means of ion-exchange and 2-thenoyltrifluoroacetone extraction, finally precipitated as the oxalate and counted for beta. The Ra^{228} is calculated from the Ac^{228} activity. Radiochemical purity can be verified by following the decay of Ac^{228} . This procedure will take as much as 6 hours for the chemistry. Because of the length of time necessary it is felt this determination cannot be considered from the economical standpoint for analysis of the evaporator effluent.

C. Strontium-90.

Two applicable methods of determining Sr^{90} are reported in the literature. The method reported by Turk (Ref. 6) can give results accurate to $\pm 30\%$ in about 0.5 hour if one omits the chemical yield determination part of this procedure. This does not include the time required for evaporating a one liter sample to 15 ml before starting the procedure. A time of 1.5 hours for the chemistry is required to give a more exacting analysis. The method involves addition of strontium carrier, repeated purification by ferric hydroxide scavenging, followed by precipitation of barium chromate as a scavenging step and finally precipitating strontium carbonate which is mounted and counted. The lower limit of activity that the method can determine is not stated. The applicability to Part 20 levels and below is inferred, but should be experimentally determined.

The second Sr^{90} method as reported by Kool (Ref. 7) can be used to determine total Sr in 5 hrs. The sensitivity is better than 0.1 MPC. Determinations of Sr for routine applications were reported in a variety of water compositions and accurate to at least 90%. The procedure does not require evaporation of initial sample to a small volume but separates Sr from a 3-liter sample by precipitating carbonates of Sr and Ba carriers. Calcium is removed by nitrate precipitation. Barium is removed by repeated barium chloride precipitations and the strontium further purified by ferric hydroxide scavenging. The strontium is finally precipitated as the carbonate, weighed, and counted for total strontium. Sr^{90} content can be defined by separating Y^{90} after sufficient ingrowth period. The advantages of this method are the separation from the bulk water sample without evaporation and the established applicability to environmental samples such as river water. In actual usage, this method may be no more lengthy than the first because of the evaporation step in the first.

D. Strontium Separation from Radium.

Although radiochemical methods are available for performing the separation and quantitative determination of Sr and Ra, these methods are somewhat time consuming, requiring as much as 13 hours. A quicker method therefore would be desirable. Ideally, a sequential separation scheme based on one pass through an ion-exchange resin column would provide a simple and perhaps a quick method. A survey of the available ion-exchange literature was undertaken.

Numerous ion-exchange schemes are described in the literature for separation of Sr (Refs. 10-19) and Ra (Refs. 4,5,12,20,21). It is possible to combine certain of these described separations schemes to provide a sequential ion-exchange method. Kraus (Ref. 12) reports a column technique for separating all the alkaline earths using an inorganic ion-exchange column of zirconium molybdate. The column material is not commercially available and, therefore, would have to be prepared in the laboratory. The absorption characteristics of cations other than alkaline earths have not been reported. Interferences would have to be established. The column operations are relatively rapid and, if the effluent solutions could be passed directly into the column without any pretreatment, a rapid determination would be possible. Developmental work would be necessary to provide a working analytical procedure. Pursuit of a development program to devise a sequential ion-exchange procedure is unwarranted for the evaluation program. The development time necessary, with possible little reduction in time of performing analyses, and economic reasons to be discussed later, of routine analysis during operation motivates this decision.

E. Cobalt-60.

One method for Cobalt reported by Schneider (Ref. 8) is designed specifically for analysis of plant waste solutions of varied composition. The method is reported as accurate to $\pm 10\%$ in the presence of total gamma activities of $1 \times 10^3 \mu\text{c/ml}$. In this procedure, Cobalt-60 is carried from solution as a mixture of insoluble cobalt (II) and iron (II) hexacyanocobaltate (III) after the aquopentamminecobalt (III) complexes have been destroyed by reduction and acid deammonation. Removal from solution is reported 99.9% complete. An average radiochemical yield of 98.1% was obtained for the complete separation procedure.

A second method can be devised based upon a method reported by Weiss and Reed (Ref. 9) for determining cobalt in seawater. Cobalt is cocrystallized with α -nitroso - β - naphthol. The precipitate is destroyed by treatment with perchloric and nitric acids, purified by ion-exchange and eluted cobalt determined spectrophotometrically by the nitroso - R - salt method (Ref. 22). This procedure could be modified to include only the α - nitroso - β - naphthol scavenging and ion-exchange purification.

The waste solution will probably contain high salt concentrations and complexing agents which might be expected to play a part in any cobalt reactions. Therefore, cobalt may be present as stable cobalt (III) ammine or other complexes, a radiocolloid or cobalt (II) ion. If these assumptions are true, the first procedure described would be recommended for cobalt analysis.

II. DISCUSSION OF COSTS FOR ROUTINE OPERATION.

In order to compare the cost of performing analyses against cost of ion-exchange columns several assumptions were used. First, it was assumed that at least 10,000 gallons of effluent could be processed through one column before exchange capacity is reached. This is felt to be a safe estimate since the ionic content of condensate from the evaporator should be very low. The cost of an ion-exchange column is \$500.00. It was assumed disposal of the ion-exchange column would cost \$200.00. Therefore, it is estimated it will cost seven cents a gallon to process the effluent. The cost of analyzing condensate from the evaporator is estimated to be about seven cents a gallon. This is based on labor alone for a GS-3 technician without added chemicals and equipment costs. Analyses performed on 300 gallon batches would require thirteen hours for the analysis time. If the analysis time could be cut to eight hours, the cost would be about five cents per gallon. But for this small reduction in cost considerable time and labor would have to be expended for development. It must be borne in mind that these analyses would increase the activity level that can be safely discharged by a factor of 60. If analyses for Ra alone were performed, the activity level that could be discharged would be increased by a factor of ten. The determination of total Ra requires an estimated four hours and the labor cost would be about 2.5 cents per gallon. Therefore, Ra analysis should be given consideration.

III. RECOMMENDATIONS.

A. If the condensate from the waste disposal unit evaporator has an activity approximately $1 \times 10^{-7} \mu \text{ c/ml}$, an analysis for total radium should be performed to determine if passage through the demineralizer is necessary. Based on time and cost consideration, analysis for other isotopes is not justifiable.

B. Laboratory testing of the proposed procedures should be performed to establish whether these are workable and applicable to this problem.

C. Development work to speed up analytical procedures by use of ion-exchange methods does not appear to be justifiable.

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